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BASES FOR THE MHTGR SOURCE TERM AND CONTAINMENT CONCEPTS

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Abstract

Significant differences in transient response and materials of construction give high temperature gas-cooled reactor (HTGRs), the potential for alternate approaches to the key issues of selection and analysis of postulated accidents, the radionuclide source-term mechanisms, containment design, and emergency planning. For the MHTGR, the siting goal is to use the U.S. Environmental Protection Agency's protection action guidelines (PAGs) for notification, sheltering, and evacuation, rather than the NRC's doses of 300 rem to the thyroid and a 25 rem whole body. This paper discusses how the design and inherent characteristic of the MHTGR lead to a radionuclide source term of prompt and delayed components. Options for the MHTGR containment design are discussed for this source term and give support to the concept of a vented, low pressure containment in comparison to the high pressure, low leakage containments characteristic of LWRs. The source term concept has been proposed to the NRC and is currently under review.

I. Introduction and Summary

Significant differences in transient response and materials of construction give high temperature gas-cooled reactors (HTGRs), in comparison with light water reactors (LWRs), the potential for alternate approaches to the key issues of selection and analysis of postulated accidents, the radionuclide source-term mechanisms, containment design, and emergency planning. The differences and alternate approaches are derived from their slow response to core heat-up events because of low core-power densities, the very high temperature the ceramic-coated particle fuel can withstand before substantial fission-product release, and the chemical inertness of the helium coolant which negates the possibility of fuel-coolant interactions. For the Modular HTGR (MHTGR), additional safety characteristics result from its design for passive reactor shutdown and passive decay heat removal. The passive means for decay heat removal is one of the subjects of this paper and is central to the retention of practically all radionuclides within the particle fuel during a postulate core heatup accident.

Early nuclear reactors were small, used crudely estimated source terms, did not have containments, and the estimated consequences were judged to be mitigated by distances to populated areas. It is historically interesting, as well as fundamental to the ensuing discussion, to note that the document entitled, "Calculation of Distance Factors for Power and Test Reactors (AEC 1962)" established the well-known "TID Source Term" which postulates a legal, quantitative release of radionuclides to the containment of 100% of the noble gas, 25% of the iodines, and 1% of the solids. For siting a given powerplant, the TID source term is used, together with the reactor containment's expected demonstrable leak rate and the metrological conditions pertinent to the site, to calculate the boundaries of the exclusion area and low population zone on the basis of allowable 2 hour and 30 day doses of radioiodine iodine to the thyroid and to the whole body dose from the total radionuclide releases. Over the years, refinement and conservatisms have entered into the calculations and work is underway at the Nuclear Regulatory Commission (NRC) and elsewhere to replace the TID source term by a mechanistically derived value and to account for severe accidents involving core melting and fuel-coolant interactions. For the MHTGR, the siting goal is to use the U.S. Environmental Protection Agency's protection action guidelines (PAGs) for notification, sheltering, and evacuation, rather than the NRC's doses of 300 rem to the thyroid and a 25 rem whole body. The PAG dose controlling the MHTGR'source term and containment analysis is a 5 rem thyroid dose at the plant site boundary.

The Fort St. Vrain HTGR and the subsequent designs for large gas-cooled reactors in the United States have used mechanistic interpretations of the TID source term and the MHTGR has departed entirely from this definition. This paper discusses the factors that enter into this mechanistic calculation and how the design and inherent characteristic of the MHTGR lead to a source term of prompt and delayed components. Options for the MHTGR containment design are discussed for this source term and give support to the concept of a vented, low pressure containment in comparison to the high pressure, low leakage containments characteristic of LWRs. The source term concept has been proposed to the NRC and is currently under review.

For the purposes of the present paper, a postulated accident involving steam ingress into the reactor caused by a steam generator tube failure is taken for determination of the source term magnitude and time sequence. Other accidents continue to be investigated, but at this time it is believed that the steam ingress sequence is the most illustrative of the prompt and delayed characteristics of the MHTGR source term. In this sequence, steam enters the reactor with the eventual result that the pressure relief valve opens and after a few relief cycles sticks open, causing the reactor primary system to depressurize. This is followed by a full duration core heatup event in which decay heat is removed passively by the reactor cavity cooling system (RCCS), to be described in the next session. Scram occurs either by insertion of absorber material or passively by negative Doppler feedback.

The prompt portion of the source term occurs during the depressurization of the primary system in a matter of minutes, while the delayed portion occurs over a period of days. The prompt source term contains radionuclides circulating with the helium, but is dominated by "liftoff" of "plated-out" radionuclides previously deposited on the cooler portions of the primary system surfaces. The delayed source term develops from the failure of a very small fraction of the fuel particles and occurs during the lengthy, core heatup phase of the accident. The fuel particle design has been described elsewhere, together with its failure modes affecting source term characteristics (Inamati, et al., 1989). This delayed radionuclide release occurs at atmospheric pressure since the prompt high pressure release has been vented to the atmosphere well before significant release of fission products from the fuel. Because the delayed release is carried out at atmospheric pressure, a high pressure, low leakage containment can be judged not to be necessary as it serves no identifiable function in this case.

II. Design and Function of the Decay Heat Removal System

The nuclear island features of the MHTGR power plant are shown in Figure 1. The reactor and heat transport components are housed in separate vessels connected by a concentric flow cross duct vessel, with all vessels housed in an underground cavity or silo. The steam generator vessel, which also contains the helium circulator and the pressure relief train, and the reactor vessel are housed in separate compartments of the cavity which, under normal conditions, do not communicate. Should an overpressure condition occur, such as could be caused by a main steamline rupture, pressure would be relieved through vent paths in the steam generator portion of the reactor building. Blowout panels connect the two

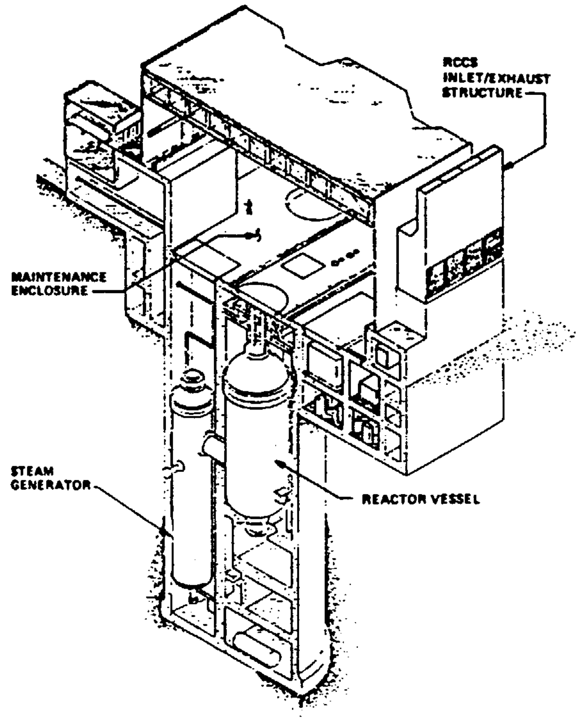

FIGURE 1: ISOMETRIC VIEW THROUGH REACTOR BUILDING

compartments. The design is such that a pressure of 10 psi in the reactor compartment could not be exceeded, which is sufficient to protect safety grade equipment in the reactor compartment from overpressure damage.

The reactor compartment contains the Reactor Cavity Cooling System (RCCS) for removal of heat transmitted to it from the uninsulated surface of the reactor vessel. This system, shown schematically in Figure 2, is a naturally convective air-cooled system of ducts and panels that is open to the environment but is closed within the reactor building. It is a safety grade structure and operates continuously. When all forced reactor cooling is lost, a low probability event, it removes decay heat fully passively at a rate sufficient to maintain fuel and vessel temperatures below acceptable limits.

The performance of the RCCS with respect to fuel and vessel temperature over time is shown in Figures 3 and 4 for reactor conditions of pressurized and depressurize, respectively. These curves are taken from independent calculations performed by Oak Ridge National Laboratory in support of NRC's on-going review of the MHTGR (Williams, et al., 1989). These calculations are in close agreement with those performed by DOE contractors. It should be noted that the maximum core and vessel temperatures are approached at about 80 hours, or 3 days, following a slow buildup.

III. Source Term Characteristics

Iodine-131, which has an 8 day half life, is considered controlling in the source term descriptions and applications discussed below. While final calculations will take into account the full spectrum of radionuclides, iodine-131 well illustrates the phenomena to be considered and is likely to be confirmed as the dominant radionuclide in the containment design basis. Table 1 (Inamate et al., 1989) characterizes and summarizes the role of this isotope with respect to its inventory, inventory location, time of release, and release mechanisms. Four inventory locations are identified; (1) circulating with the helium coolant, (2) plated out on primary system surfaces (3) associated with defective fuel particles, and (4) contained within standard fuel particles. Except for the standard fuel particle inventory, the inventories given are nominal and subject to uncertainties being addressed in the technology development program. These, and the uranium dicarbide inventory within the defective fuel particles, which is released rapidly under hydrolysis conditions, form the prompt source term and are released in a matter of minutes following failure of a relief valve to close. Although uncertainties exist, it is evident that the prompt source term will be sufficiently small that it can be vented from the reactor building. If subsequent research determines that the prompt source term is larger than currently predicted, it can be vented through a filter on the relief train to meet goal release quantities. It is important to note that after the prompt release there is no further pressure driving force for subsequent transport of radionuclides.

The delayed source term is based on the inventory of contaminated and defective fuel and develops over a period of hours to days as the temperature of the core elevates with time as previously described. By contaminated fuel, it is meant that the minute portions of uranium not

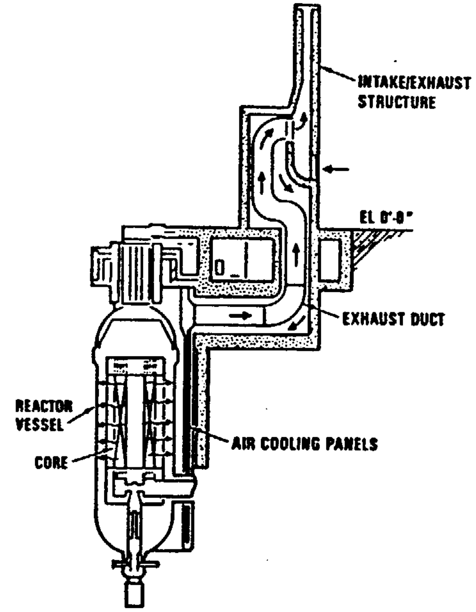

FIGURE 2: PASSIVE REACTOR CAVITY COOLING

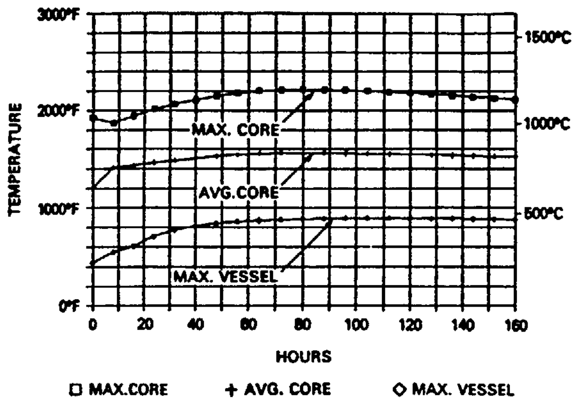

FIGURE 3: PRESSURIZED CONDUCTION COOLDOWN WITH RCCS TEMPERATURES OF CORE AND VESSEL VS. TIME (MORECA REFERENCE CASE)

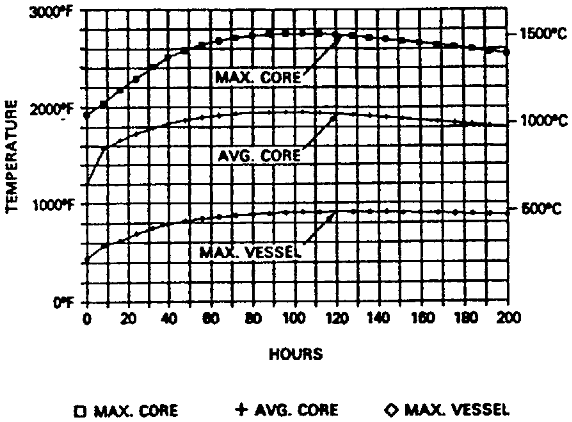

FIGURE 4: DEPRESSURIZED CONDUCTION COOLDOWN WITH RCCS TEMPERATURES OF CORE AND VESSEL VS. TIME (MORECA REFERENCE CASE)

encapsulated in the particles during manufacture remain to contaminate graphite regions exterior to the coating barriers. Defective fuel is that fuel which contains at manufacture or during operation coating fractures or weaknesses which effectively negate the coating barriers to fission product release during core heatup. As this release is proportional to temperature, the inventory is released over a period of hours and days and is expected to roughly follow the core temperature rises given in Figures 3 and 4. As releases from the delayed source term occur effectively at atmospheric pressure, these releases are subject to different transport phenomena than those associated with high pressure, low leakage containments.

IV. Phenomenological and Design Development Needs

Laboratory testing and operating reactor experience with ceramic particle fuel in both the United States and Germany, together with modern design and analytical techniques, give good confidence in the expected source term behavior and in the performance of the reactor cavity cooling system. A program of testing and computer code verification and validation is being developed.

For RCCS, performance validation by coordination with the New Production Reactor (NPR) version of the MHTGR is expected if this design is selected for the tritium production application. If the MHTGR is not selected, the following other options being considered are: (1) full reliance on computer models and existing correlations with confirmation by testing during start-up and power ascension, (2) a cooperative program with the PRISM liquid metal reactor which uses a similar passive heat removal system, and (3) development of an international, IAEA cooperative research program (CRP) on decay heat removal.

The items of fuel integrity and reliability, fission product transport, and the effects of water ingress are three subjects for which extensive development and confirmation testing are planned. We believe, however, our source term model, as summarized in Table 1, is sufficiently valid for the present stage of our program as it is based on related U.S. and German experiments, tests and reactor operations (ACRS 1992). A detailed description of our specific plans for development and confirmation of the source term is beyond the scope of this paper, but we hope to involve international activities in achieving our goals. Major radionuclide transport quantities to be established by experiment are the hydrolysis effects on uranium oxide fuel and the magnitudes of lift-off and wash-off during rapid depressurization.

TABLE 1. MHTGR SOURCE CHARACTERIZATION FOR DOMINANT NUCLIDE CONTRIBUTING TO THYROID DOSE

RADIONUCLIDE SOURCE CHARACTERIZATION	INVENTORY (Ci I-131)	TIMING OF RELEASE	RELEASE MECHANISMS	
			FROM CORE	FROM PRIMARY CIRCUIT
1) Circulating	0.02	minutes	--	Flow (No Depress.)
2) Plateout	20.0	minutes	--	Flow (No Depress.) Moisture (Water Ingress)
3 Initially Defective Particles				
a) Contamination	93	hours - days	Temperature (Loss of Forced Cooling)	Flow (No Depress.)
b) Defects	465	hours[1]-days	Temperature (Loss of Forced Cooling)	Flow (No Depress.)
			Moisture (Water Ingress)	Flow (No Depres.)
4) Standard Particles	$9x10^{+6}$	> days	Temperature (No Event Identified)	--

1. Approximately 7% (the UC2 fraction) of the inventory in non-intact particles is subject to release in a time frame of minutes under hydrolyzing conditions encountered in rare MHTGR accidents.

We are studying several design options to achieve a level of reduction and attenuation of radionuclides to assure that the design will meet the U.S. PAG goals at the site boundary with margin. These options are enlargement of the site boundary, elevated release through a stack, and the use of filters and a more tortuous path within the reactor building to the vented locations. We are giving our current attention to the possible use of filters on the relief train, schematically illustrated in Figure 5, on the reactor building itself, Figure 6, and reduction in the reactor building leak rate from 100 to 5 per cent per day. We have not yet selected filter types, locations, and other means to address the options being considered.

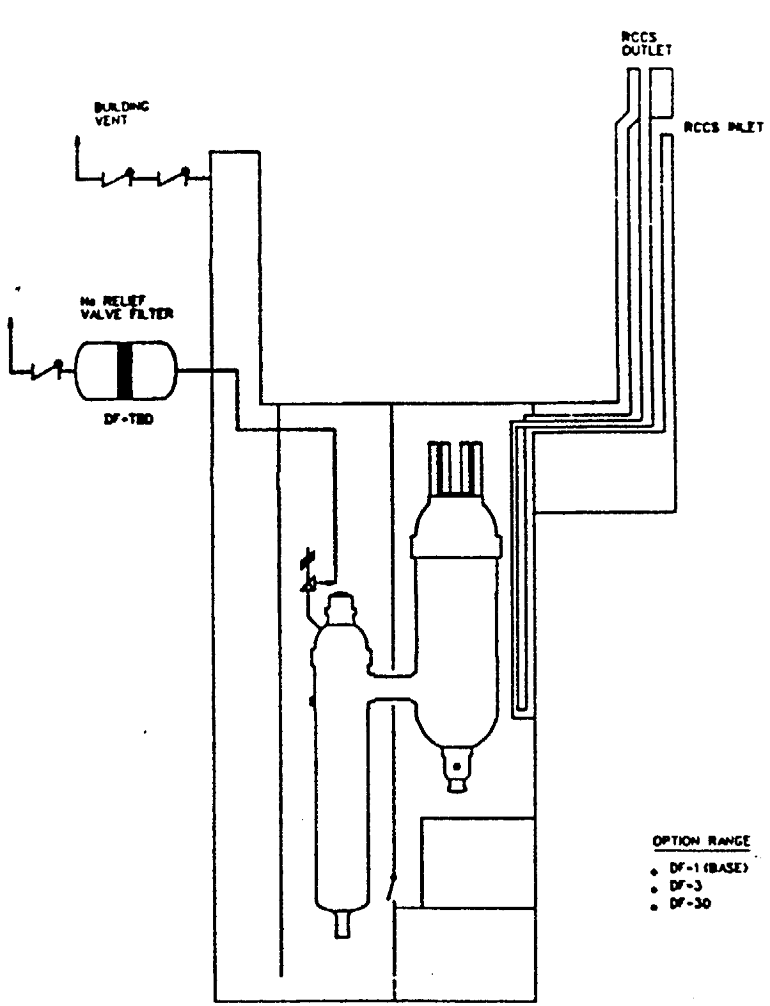

FIGURE 5: HELIUM RELIEF VALVE FILTER TRAIN DESIGN OPTION

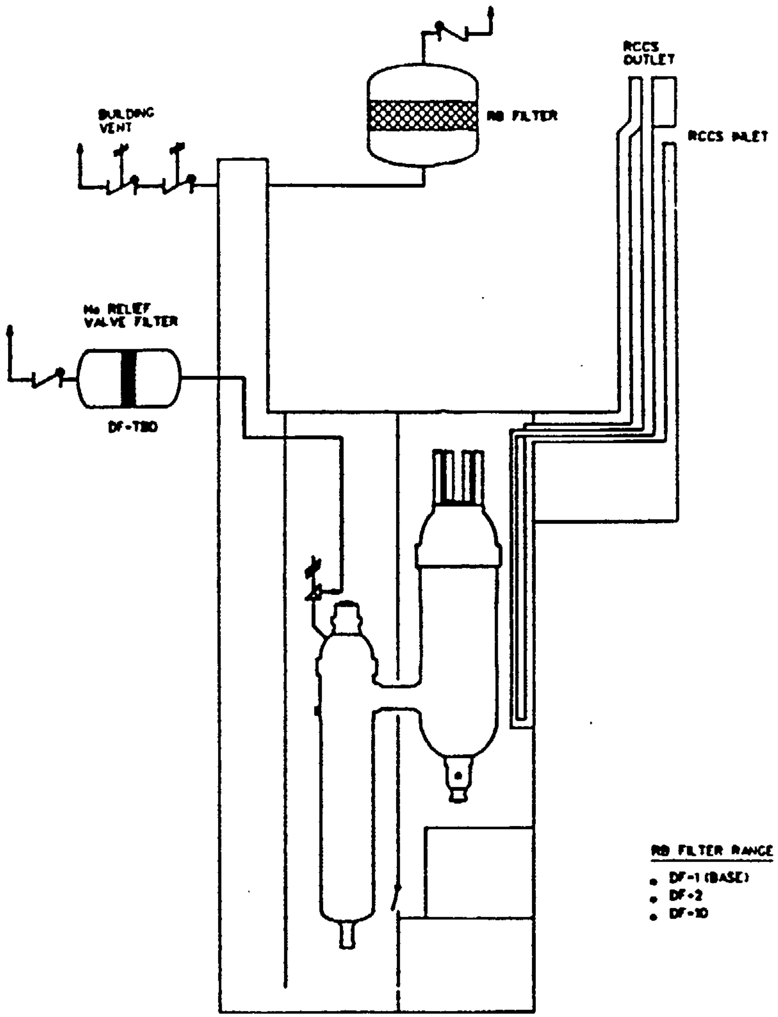

FIGURE 6: REACTOR BUILDING FILTER TRAIN DESIGN OPTION

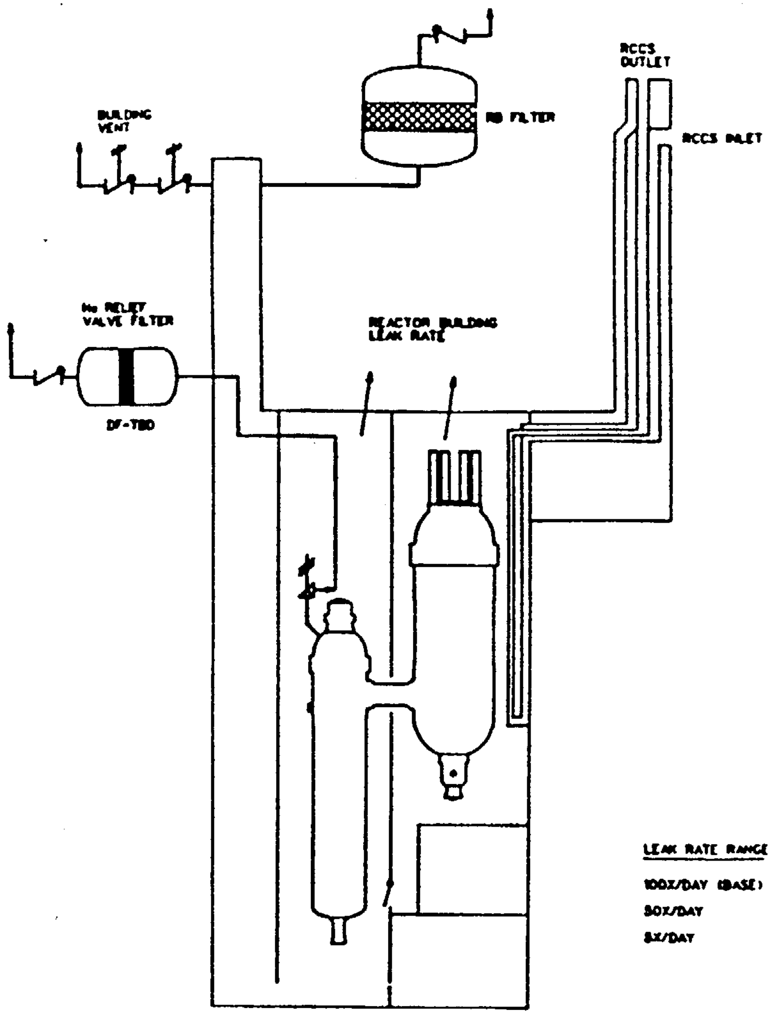

FIGURE 7: REACTOR BUILDING LEAK RATE DESIGN OPTION

As stated, we are designing so that the PAG goals can be met with margin, which is good engineering practice and also provides for licensing uncertainties. We anticipate possible imposed conservatisms, although do not believe them to be necessary, to include an increase in the magnitude of water ingress, increased overall margins for "defense-in-depth" and general prudence considerations, and possible reversion to more traditional, non-mechanistic view points regarding the source term definition. Overall, we believe we have an approach to the containment that is robust, in keeping with a properly established source term, and should be acceptable to regulators and the public in general.

V. Conclusions

The source term and containment concepts for the MHTGR that have been identified in this paper are consistent with U.S. and international past HTGR reactor operations, more recent fuel development findings, and current design and analysis studies. We are planning to evaluate containment options and establish our reactor building design over the forthcoming year. Our work is being based on the following:

1. Development and use of a mechanistic source term for the MHTGR, as compared to use of an arbitrary source term. We believe that this is a superior approach to achieve reactor safety and licensing goals.

2. Containment concerns for the MHTGR are best addressed by including in the containment system a vented, low pressure reactor building that recognizes that the MHTGR source term has distinct prompt and delayed components.

3. The design options outlined herein can be engineered with no problem of feasibility, although research and development activities are needed to establish margins and to optimize the design.

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